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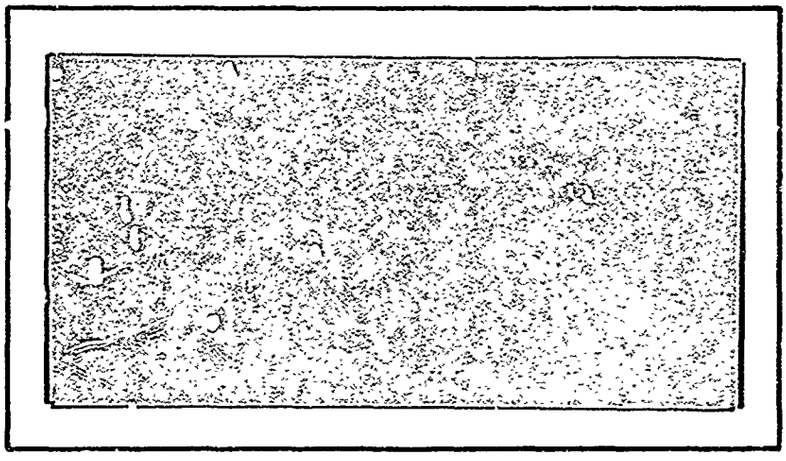

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OLD BARNYARD: A CROSS SECTION CODE FOR SCHOOL USE

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PREFACE

It is my policy to teach nuclear reactor physics as a problem oriented course. Such problems generally require, as input, group cross sections and cross section related constants such as the diffusion coefficient or neutron age. However, these problems, unlike their counterparts in the real world, usually involve only one or a few energy groups. This few group distinction has always made the acquisition of group cross sections a particularly frustrating experience. One borrows an experimental value from here, a calculated value from there, and combines them with some fast cross sections collapsed by hand from a many group set (which inevitably doesn't contain all the elements of interest). Although sophisticated methods are always available, data for one problem of a 10 problem assignment rarely warrants the effort to obtain them; nor does the accuracy required in a school problem warrant their sophistication.

The code described in this report is an attempt to provide reasonably good cross sections from a single, easily used, and unsophisticated (therefore inexpensive) source. The code is written for the IBM 1620, a digital computer, installed in 146 (55%) of the 268 schools listed in the 1964 Oak Ridge Institute of Nuclear Studies report, EDUCATIONAL PROGRAMS AND FACILITIES IN NUCLEAR SCIENCE AND ENGINEERING.

Robert H. Hansen carried out the moments reduction study as his M.S. thesis. Ernest Park Sims added the thermal portion of the code in conjunction with his M.S. thesis research, and put the code in essentially its present form. In addition, acknowledgment is due Phillip A. Terry for the inclusion of fission cross sections.

Finally I feel compelled to make some comment about the name OLD BARNYARD. It was originally coined as a joke; however, I find it is a name which the student remembers and more importantly associates with cross sections and age. I have yet to have a student ask "What was the name of that code which generates cross sections?"

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Typed by Miss Shari L. Johnston

OLD BARNYARD: A CROSS SECTION CODE FOR SCHOOL USE

ABSTRACT

This interim report describes a digital computer code which calculates few group fast and thermal nuclear cross sections and constants. It performs these calculations using the moments method for fast flux and Wigner-Wilkins method for thermal flux. The code is designed for use on the IBM 1620 computer and is intended primarily for use by students and professors in support of classroom assignments.

INTRODUCTION

Reactor physics instructors who assign their students criticality or associated problems are continually faced with the problem of providing group cross sections and cross section related constants such as diffusion coefficient and age. The alternative to providing such constants is to require each student to find or calculate his own values. This alternative is a guarantee of considerable effort, of doubtful educational value, on the part of the students and of considerable variation of final answers for the instructor to evaluate. Standard practice in most first year courses in reactor physics is to use a Maxwell Boltzmann distribution to determine thermal group constants plus experimental values as available, and to borrow epithermal and fast group values from some published set such as those in ANL 5800[1]. Occasionally an instructor will have the resources to obtain fast group cross sections from a moments code such as GAM[2]. However, since the instructor usually wants to use only a few groups for learning purposes, this last option is inefficient in time and dollars. More likely he will

end up averaging or collapsing someone else's published 16-30 group set.

This use of the Maxwell Boltzmann distribution for thermal cross sections is usually adequate, but the crude collapsing of a many group set for fast cross sections leaves something to be desired. As fast reactor calculations become more and more important in the classroom, this problem is amplified. Further when cross sections are specified in this manner, one can almost guarantee a mismatch in values such as $L^2 \neq D/\Sigma_a$. All of this uncertainty is very disconcerting to the beginning student.

The computer code described in this report is the result of an attempt to provide a simple easily used code to generate few group cross sections and constants for school use. The code calculates both fast and thermal cross sections and constants such as age and diffusion length. It is designed for execution on the IBM 1620 computer where it requires about 15 minutes for execution. The cross sections are found by calculating the energy dependent flux in the infinite homogeneous mixture of the isotopes or compounds specified. The input cross section data is obtained from an eleven group library of fast cross sections plus 2200 m/sec thermal values. The energy dependent flux is calculated by the neutron transport, moments method for above thermal energies and by a Wigner-Wilkins calculation for thermal energies.

The moments calculations based on an 11 group input was arrived at after a systematic reduction in the fine group structure, until on acceptable compromise between accuracy and amount of calculation

was found at 11 fine groups. The above-thermal calculations were further simplified by computing resonance absorption with a resonance integral taken from the formula given by Murray[3] and other texts. The Wigner-Wilkins spectra is normalized to the moments spectra at 1.125 ev. The details of these calculations are given in section titled "Theory."

The code itself is described in the section titled "The Code" along with simplified operating instructions and output samples. Again the unique feature of this code is that unlike the moment codes and thermal codes now available it can be executed on a small machine (IBM 1620) in about 15 minutes or roughly one dollar of computer time.

THEORY

In this section the methods of solution used for various important calculations in the code are presented. Derivations are cursory and less important or routine calculations are omitted. An attempt is made, however, to cite appropriate references where the interested reader may find a detailed discussion of the subject.

Epithermal Age and Flux

The calculation of epithermal age and flux is made using the moments method; i.e., the moments of the Fourier transformed P_1 approximation to the Boltzmann neutron transport equation.

The derivation of the equations involved is straightforward but long. R. H. Hansen[4] has treated them in great detail. The resulting finite-difference equations to be programmed are

$$\alpha_n \chi_n + \sum_{j=1}^{n-1} \frac{\sigma^o_{s\, j\to n}}{\sigma_{Rj}} \alpha_j$$

$$\Phi^1_{1n} = \frac{\alpha_n}{3\sigma_{mgtn}\sigma_{Rn}N^2} + \frac{1}{\sigma_{mgtn}} \sum_{j=1}^{n-1} \sigma^1_{s\, j\to n} \Phi^1_{1j} \qquad (1)$$

$$\tau_n = \frac{\Phi^1_{1n}}{\alpha_n} + \frac{1}{\alpha_n} \sum_{j=1}^{n-1} \frac{\sigma^o_{s\, j\to n}}{\sigma_{Rj}} \alpha_j \tau_j$$

where $\alpha_n = N\sigma_{Rn}\Phi^o_{on}$

$$\sigma_{mgtn} = \sigma_{tn} - \sigma^1_{s\, n\to n}$$

and N = total nuclei density, nuclei $cm^{-3} \times 10^{-24}$

σ_{Rn} = microscopic removal cross section for the n^{th} group, barns

σ_{Tn} = microscopic total cross section for the n^{th} group, barns

$\sigma^{o}_{s\ j\to n}$ = zeroth harmonic of the microscopic elastic scattering cross section from group j to group n, barns

$\sigma^{1}_{s\ j\to n}$ = first harmonic of the microscopic elastic scattering cross section from group j to group n, barns

Φ^{p}_{mn} = p^{th} derivative of the m^{th} moment of the Fourier transformed flux in group n

τ_n = neutron age to the lowest energy of the group n, cm^2

χ_n = fission source in group n

In addition to calculating age, the above equations generate relative group fluxes through the Φ^{o}_{o}'s. These fluxes and age values are calculated down to lethargy 17 (0.414 ev). All values of $\phi(u)$ are normalized to $\phi(16) = 1$ in order to join with the thermal flux at the same lethargy value.

Resonance Escape Probability

The calculation of resonance escape probability follows the treatment given in Murray's text[5]. The scattering per resonance atom is calculated and the value obtained is used to select an empirical expression for an effective resonance integral. The empirical equations used are taken from Isbin[6] and are reproduced below:

For U^{238}

$$(RI)_{eff} = 2.69 \left(\frac{\Sigma_s}{N^{238}}\right)^{0.471} \qquad 0 \le \frac{\Sigma_s}{N^{238}} \le 4000 \qquad (2a)$$

$$\ell n(RI)_{eff} = 5.64 - \frac{163}{(\Sigma_s/N^{238})^{0.65}} \qquad \frac{\Sigma_s}{N^{238}} > 4000 \qquad (2b)$$

$$RI_{eff} = 280 \text{ barns} \qquad \Sigma_s = \infty \tag{2c}$$

For Th^{232}

$$(RI)_{eff} = 8.33 \left(\frac{\Sigma_s}{N^{232}}\right)^{0.253} \qquad 0 \le \frac{\Sigma_s}{N^{232}} \le 4500 \tag{3a}$$

$$(RI)_{eff} = 70 \text{ barns} \qquad \frac{\Sigma_s}{N^{232}} > 4500 \tag{3b}$$

where Σ_s = macroscopic scattering cross section for the mixture over the resonance region, cm^{-1}

N^A = nuclei density of nuclide A, cm^{-3}

The value of the resonance escape probability is then found from

$$p = \exp\left[-(RI)_{eff} N^A / \xi \Sigma_s\right]$$

where $\xi \Sigma_s$ is for the mixture.

This value of p is then introduced into the equations (1) by multiplying σ_{R_8} by p (Resonance takes place in group 8). This artifice has the effect of reducing the downscatter to group 9 by appropriate factor without violating neutron conservation within the equations.

Should the user so desire, he may input his own value of $(RI)_{eff}$ to be used in the code.

Absorption Parameter

The absorption parameter is defined for a single nuclide by

$$\Delta = \frac{2A\sigma_a(E)\sqrt{E/kT}}{\sigma_s(E)} \tag{4}$$

where A = mass of the nuclide, amu

$\sigma_a(E)$ = microscopic absorption cross section at energy E, barns

$\sigma_s(E)$ = microscopic scattering cross section at energy E, barns

k = Boltzmann's constant

T = temperature, °Kelvin

Since Δ is used only for thermal range calculations, $\sigma_s(E)$, is assumed constant and $\sigma_a(E)$ is assumed to vary as 1/v. Then

$$\Delta = \frac{2A\sigma_{ao}}{\sigma_s} \tag{5}$$

where σ_{ao} = microscopic absorption cross section at energy kT, barns

To find Δ for a mixture of nuclides the formula

$$\Delta = \frac{1}{(1-f)} \sum_i \frac{\Sigma_T^i \Delta^i}{\Sigma_T^i} \tag{6}$$

is used where

f = thermal utilization

Σ_T^i = total macroscopic thermal cross section for the ith nuclide, cm^{-1}. This technique was not found in the literature, except for the thermal utilization factor which is suggested by Weinberg and Wigner[7].

Thermal Spectrum

The thermal spectrum is calculated by solving the Wilkins equation for the distribution of neutrons in an infinitely heavy Maxwellian gas. The equation is [8]

$$xN''(x) + (2x^2-1)N'(x) + (4x - \Delta)N(x) = 0 \tag{7}$$

where x = ratio of neutron velocity to that at energy kT

N(x) = number of neutrons at normalized velocity x and primes denote differentiation with respect to x.

The equation is solved by setting

$$N(x) = x^2 e^{-x^2} M(x) \tag{8}$$

which converts it to

$$xM''(x) + (3-2x^2) M'(x) - \Delta M(x) = 0 \tag{9}$$

This equation is then solved by the power series method.

The result is

$$M(x) = \sum_{n=0}^{\infty} a_n x^n \tag{10}$$

where

$$a_1 = a_0 \frac{\Delta}{3}$$

$$a_2 = a_0 \frac{\Delta^2}{24}$$

$$a_n = \frac{1}{n(n+2)} [2(n-2)\, a_{n-2} + \Delta\, a_{n-1}] \tag{11}$$

In the code the value of $N(x)$ and $x^2 N(x)$, which is $\phi(u)$, are calculated at every 0.25 lethargy interval beginning at $u = 23$ and continuing to $u = 16$. The calculation employs equation (10) and adds terms until contributions of additional terms are less that 10^{-6} relative value. All thermal fluxes are normalized to $\phi(16) = 1$ in order to match the epithermal fluxes from the moments calculation.

Thermal Constants

The thermal range in the code is arbitrarily defined as the interval 0 to 1.125 ev which corresponds to ∞ to 16 on the lethargy scale. All thermal constants are calculated for this interval.

Most Probable Neutron Velocity. The most probable thermal neutron velocity is found by locating the peak of the N(x) vs x curve generated in the thermal flux calculation. Since values of N(x) are calculated at intervals of 0.25 lethargy rather than continuously the value of v_p found is not exact.

Average Neutron Velocity. The average neutron velocity is found from the formula

$$\bar{x} = \frac{\int_0^{6.67} x\, N(x)\,dx}{\int_0^{6.67} N(x)\,dx} \tag{12}$$

where x is again the normalized velocity and 6.67 is that value of x corresponding to lethargy 16. The integral over N(x) is calculated analytically by integrating the Wilkins equation. The integral over xN(x) is found by the trapezoidal rule during the thermal flux calculation. Once $\bar{x}$ is found the most probable neutron velocity is given by

$$v_p = \bar{x}\sqrt{1.65 \times 10^4\, T} \tag{13}$$

where T is again system temperature.

Thermal Diffusion Length Squared. The value of thermal diffusion length squared is calculated from the equation [9]

$$L^2 = \frac{\Sigma_s\, \bar{x}}{3\, \Sigma_T^2\, \Sigma_{ao}} \tag{14}$$

$$L^2 = \frac{\sigma_s\, \bar{x}}{3\sigma_{ao}\,[(\sigma_s + \sigma_{ao}\sqrt{x})N]^2} \tag{15}$$

where L = thermal diffusion length, cm
and where $\sigma_{ao} = [x\sigma_a(x)]_{x=x_p}$, x_p is most probable value.

Here the cross sections and nuclei density are input data to the program and $\bar{x}$ is calculated as described earlier.

Migration Area. The total migration area for a neutron is found by adding the moments method age to lethargy 16 to the value of L^2 for the lethargy interval ∞ to 16.

Thermal Cross Sections. The 2200 m/sec cross sections for appropriate nuclides are input data to the code. These values are adjusted by the code to account for deviation of the thermal spectrum from Maxwellian. In particular the proper absorption cross section for the thermal group is

$$\bar{\sigma}_a = \frac{\sigma_{ao}}{\bar{x}} \tag{16}$$

and the transport cross section is

$$\sigma_{tr} = \bar{\sigma}_a + \sigma_s(1-\bar{\mu}) \tag{17}$$

where the term $(1-\bar{\mu})$ is also input data.

Cross Section Collapsing

Cross section libraries for the code have a twelve group structure; eleven epithermal and one thermal. Only the epithermal groups may be collapsed. The thermal group is always left intact and is output in the same manner regardless of the number of epithermal groups.

The eleven epithermal fine groups may be collapsed into any number of broad groups from one to eleven. The boundaries of the broad group are restricted to those values which are boundaries of the fine groups.

Broad group cross sections are generated from the formula

$$\sigma_n^{BG} = \frac{\sum_i \phi^i \sigma_n{}^i}{\sum_i \phi^i} \tag{18}$$

where σ_n^{BG} = microscopic cross section of the n^{th} kind for a broad group

ϕ^i = total flux in the ith fine group

σ_n^i = microscopic cross section of the n^{th} kind for the ith fine group

An exception to the method of equation (18) is the calculation of the transport cross section. Here the use of σ_{tr} in a reciprocal manner through the diffusion coefficient is anticipated so that

$$\frac{1}{\sigma_{tr}^{BG}} = \frac{\sum_i \phi^i \frac{1}{\sigma_{tr}{}^i}}{\sum_i \phi^i} \tag{19}$$

is used.

A number of different cross sections are output by the code and the interrelationships which exist among them is not always obvious. The following explanation of more obscure relations may prove useful to the user.

Total Transfer. The group to group total transfer cross section is that value which when multiplied by the losing group flux gives the rate of total transfer from the losing group to the gaining group. Thus total transfer from j to k consists of the sum of inelastic, elastic (P_o), and twice n-2n from j to k. Note that elastic (P_1) which is $\bar{\mu}$ times elastic (P_o) does not contribute to total transfer.

Removal. The removal cross section is applied only to a broad group and is that value which when multiplied by its broad group flux gives the rate of removal of neutrons from the group. Thus the removal cross section is the sum of absorption, inelastic removal, elastic (P_o) removal, and one times the n-2n removal cross sections. Note that in-group cross sections are not included as they do not remove neutrons from the group.

THE CODE

The digital computer code which carries out the calculations described in the previous section is written in the Kingston version of Fortran II[10] (sometimes called "Kingstran"). The code consists of two chains which must be run sequentially. The first chain computes the energy dependent flux, both fast and thermal. The output of this chain consists of the group flux for each of the 11 fine fast groups plus 9 thermal values at half lethargy increments as well as age, diffusion length squared, resonance integral parameters, and average and most probable neutron velocities in the thermal region. In addition the chain one output includes a plot of the flux as a function of lethargy over the entire range of interest. The second chain collapses the cross sections to the desired group structure using the flux calculated in chain one. The operator may choose between microscopic and macroscopic cross sections for chain two output. A listing of the source decks for chains one and two is contained in Appendix A. The source decks shown there have been executed on an IBM 1620 with 40K memory, card input-output, floating divide, indirect address and the usual additional instructions(TNS,TNF, MF).

Operating Instructions

1. Set all console switches to PROGRAM except the Parity switch is set to STOP.

2. Place special Old Barnyard short subroutines in the read hopper. Press RESET, LOAD.

3. When last card of subroutines is reached, the console will give READER NO FEED light. Press READER START and read in last card.

4. Remove subroutines from the out hopper. Place Old Barnyard chain one object deck in the read hopper. Start the PUNCH. Place data cards for chain one on top of chain one object deck in read hopper. Press LOAD.

5. Chain one program will run automatically. When last data card enters reader, console will give READER NO FEED light. Press READER START to read in last card.

6. Console typewriter will indicate the beginning of each section of the code. At the end of the thermal calculation the typewriter will type "SET SWITCH 1 ON TO PRINT OUT SPECTRUM. PRESS START." If a plot of the total neutron spectrum is desired set switch 1 ON and press START. If no spectrum plot is desired, set switch 1 OFF and press START.

7. At the end of chain one, the console typewriter will type "END OF CHAIN 1. TO COLLAPSE CROSS SECTIONS LOAD CHAIN 2." If no cross sections are desired program is finished. If cross sections are desired place Old Barnyard Chain 2 object deck in the read hopper. Place data cards for Chain two on top of it. Press RESET, LOAD. Chain two will run automatically. When last data card enters reader, console will give READER NO FEED light. Press READER START to read in last data card. Remove Chain two and data dards from out hopper.

8. When chain two is finished the typewriter will write "END OF PROGRAM." Remove output cards from output hopper. It is not necessary to press NON PROCESS RUN OUT.

9. List output answer cards on the 407 lister. Put switch number 4 at the right end of the machine UP. This causes automatic advance of a fresh sheet of paper for each new reaction of the output answers. If the paper advances at unwanted times (as it sometimes does) put switch 4 DOWN. (The best plan is to get switch 4 UP for chain 1 output, DOWN for chain 2 output).

10. To run a second problem go back to step (1) and repeat.

NOTE: If anything should go wrong during a run (such as an erroneous data card, or a I/O error, or a SKIP CHECK) the particular chain in use may be restarted by the following procedure:

1) Press STOP

2) Remove remaining data cards, if any, from the read hopper.

3) Clear the reader of any internal data cards by pressing NON-PROCESS RUN OUT.

4) Replace corrected data cards in the read hopper.

5) Press RESET, INSERT.

6) Type on the console typewriter the numbers 4900936.

7) Press RELEASE, START or hit the R-S key on the typewriter.

8) Press READER START to read in data cards

9) Program will run.

Preparation of Input Data, Chain One.

The following data cards are required to operate OBY Chain one. Unnecessary decimal points may be omitted on all cards.

Card No. 1:

This card contains 65 columns of alphameric data of the user's choosing. Identification, problem title, date, user's name, etc. are possible information to be placed on this card. The data on this card will be typed out on the console typewriter and will be punched at the head of the output data.

EXAMPLE: Checkout of OBY VIII for Carbon. 25 May 66 Wetzler.

Card No. 2:

This card carries five (5) numbers as follows, in any desired format:

a) A number between two (2) and twenty-one(21) indicating the source of neutrons which the user wants to use. The meaning of these numbers is shown below:

Number	Source
2	U^{235} fission (Cranberg Spectrum)
3	U^{233} fission
4	Pu^{239}
5	Pu^{241}
6	Cf^{252}
7	Pu-Be (Whitmore-Baker)
8	Pu-Be (Cochrane-Henry)
9	Ra-Be (Hill)
10	The user's own source (See Card No. 4)
11-21	A unit source in group 1-11 respectively

EXAMPLE: 15 = a unit source in group 5.

b) A second number indicating the number of nuclides to be used in the problem.

EXAMPLE: 2 for water (H and O)
2 for BeO (Be,O)
2 for C_2H_6 (C,H)
3 for an Al_2O_3-H_2O mixture (Al,O,H)
1 for pure carbon (C)

c) A number giving the total nuclei density for the problem in atoms per cm^3 times 10^{-24}.

EXAMPLE: for carbon 0.08025
for H_2O 0.1005
(Note: See ANL 5800 page 30)

d) A number giving the temperature of the system in degrees Kelvin.

e) A number indicating whether or not a resonance escape calculation should be performed; as below:

1 = Yes
2 = No

Card No. 3:

This "card" is a small deck of cards containing the source data for the program. It is included as a part of the input library and normally requires no preparation on the part of the user. A listing of the sources in the authors' library is shown in Figure 1.

Card No. 4:

This card is required only if the first number on card no. 2 was "10"; i.e., only if the user wants to read in and use his own source. The card will contain eleven numbers in any format representing the source in each of the first eleven groups of the

```
.5 1. 1.5 2. 2.5 3. 5. 8. 12. 16. 17.
THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).
2 .023023 .10824 .21044 .23139 .18048 .11483 .12439 .0072094 .0 .0 .0
THE SOURCE USED IS U233 FISSION.
3 .021495 .10472 .20811 .23202 .18252 .11674 .1267  .0073845 .0 .0 .0
THE SOURCE USED IS PU239 FISSION.
4 .024567 .11047 .21062 .22995 .17917 .11412 .1239 .0071986 .0 .0 .0
THE SOURCE USED IS PU241 FISSION.
5 .026852 .11615 .215 .22949 .11077 .1190 .0068574 .0 .0 .0
THE SOURCE USED IS CF252 FISSION.
6 .046142 .14786 .2292 .21936 .15472 .09531 .099137 .0055827 .0 .0 .0
THE SOURCE USED IS PO-BE (WHITMORE-BAKER).
7 .2342 .357 .261 .1025 .036 .0085 .0 .0 .0 .0 .0
THE SOURCE USED IS PO-BE (COCHRANE-HENRY).
8 .257 .369 .2514 .079 .028 .010 .0 .0 .0 .0 .0
THE SOURCE USED IS RA-BE (HILL).
9 .239 .365 .189 .109 .065 .033 .0 .0 .0 .0 .0
THE SOURCE USED IS YOUR OWN AS SHOWN BELOW.
10. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0.
THE SOURCE USED IS A UNIT SOURCE IN GROUP NO.
10 .0 .0 .0 .0 .0 .0 .0 .0 .0 .0 .0
```

Code:
First line: Upper lethargy limits of the 11 fine groups.
Second line: Identification
Third line: Identification number followed by the percent of the source found in the 11 fine groups, in the order $S_1, S_2, \ldots, S_{11}$

Figure 1: Input Library, Sources

program group structure.

Card No. 5:

This card is necessary only if the fifth number on card no. 2 was 1; i.e., if a resonance escape calculation is to be performed. The card will contain three (3) numbers as follows: (any format)

a) A number indicating whether the resonance nuclide present is U^{238}, Th^{232}, or W (Tungsten) as below:

1 = U or W
2 = Th
1 = any mixture of the above 3

b) A number giving the nuclei density $\left[\frac{\text{atoms}}{\text{cm}^3} \times 10^{24}\right]$ of the resonance nuclide(s) in the problem.

c) A number indicating whether the effective resonance integral is being supplied by the user, or whether it is to be calculated by the program, as below:

0 = to be calculated

non-zero = value of RI_{eff} to be used.

Card No. 6:

This card contains five pieces of data as follows:

a) 15 columns of alphameric identification to identify the n^{th} nuclide of the problem. This is optional, for the user's use only, and may be left blank.

b) A number giving the nuclei density of the n^{th} nuclide in $\left[\frac{\text{atoms}}{\text{cm}^3} \times 10^{-24}\right]$ in the problem.

c) A number indicating whether the nuclei is a moderator, fuel, or other, as below:

1 = moderator
2 = fuel
3 = other

d) A number indicating whether micro - or macroscopic cross sections are to be output for this n^{th} nuclide. This number is necessary only if cross section averaging is to be performed. (Chain two of the program). The number should be:

1 = micro
2 = macro

e) A number indicating whether fission cross sections are to be included:

1 = yes (fission)
2 = no (fission)

Card No. 7:

This "card" is a deck of cards bearing the 11 group plus thermal cross sections of the n^{th} nuclide. This deck is selected from the program library and requires no preparation on the part of the user. Some examples of the author's library are shown in Figures 2 and 3.

Cards 6 and 7 are required for each nuclide in the program. The ordering of nuclides is completely arbitrary. Figure 4 shows the arrangement of the entire input deck for Chain one.

Notes on Chain One data:

a) The total nuclei density on the first data card should be the sum of the individual densities on the separate card no.6's.

b) The nuclei density of the resonance nuclide on card number 5 should agree with that on card no. 6 for the absorber

```
   CARBON(a)      62(b) 0(c) 1(d) .158(e) 4.81(f)   .004(g) .02(h) 4.806(i)   .9444(j)      (k)
2.0000(l)      6.0000(m)  [0.0000       3.5757E-02  1.0514E-01  7.8474E-02
4.6774E-02     2.7711E-02   3.1662E-02  0.0000      0.0000      4.0467E-03
8.0933E-03     4.6132E-03   5.5135E-03  0.0000](n)  1.1000E+01(o) 1.0000(p)
[5.5748E-01    1.1047       3.2908E-01 -3.3102E-01  9.8070E-01  1.3262
5.8681E-01    -5.6251E-01   1.0714      9.6768E-01  7.1394E-01 -8.5887E-01
1.2910         1.0688       6.5028E-01 -4.7099E-01  1.7890      1.5656
8.1709E-01    -5.6305E-01   2.1883      1.8060      9.6579E-01 -7.0608E-01
3.7440         1.1643       3.2552E-01 -2.4896E-01  4.3996      9.8826E-01
2.4442E-01    -2.1432E-01   4.5213      9.4884E-01  1.8565E-01 -1.6434E-01
4.5242         9.4954E-01   1.8577E-01 -1.6455E-01  3.9669      1.4432
7.4309E-01    -6.5822E-01](q)
```

Code:

a) Identification of isotope (element). 1st 15 columns.
b) Number of fast cross section enteries below (beginning on 2nd line)
c) Epithermal-fast absorption. 0 = negligible; 1 = not negligible
d) Type of inelastic scatter: 1 = inelastic; 2 = (n,2n); 3 = both; 4 = neither.
e) Thermal average logarithmic energy decrement, ξ
f) Thermal, total, microscopic cross section, σ_{tot}
g) Thermal, absorption, microscopic cross section, σ_a
h) Thermal, absorption parameter, Λ
i) Thermal, scattering microscopic cross section, σ_s
j) Thermal $(1-\bar{\mu})$ where $\bar{\mu} = 2/3A$
k) Thermal value of the product of neutrons per fission and the microscopic, fission cross section, $\nu\sigma_f$
l) Number of groups from which neutrons inelastically scatter
m) Number of groups to which neutrons inelastically scatter
n) Microscopic, fast, inelastic, cross sections in the following order:

$$\sigma^{in}_{1\to1} \; ; \; \sigma^{in}_{1\to2} \; ; \; \ldots \; \sigma^{in}_{1\to7}$$

$$\sigma^{in}_{2\to2} \; ; \; \sigma^{in}_{2\to3} \; ; \; \ldots \; \sigma^{in}_{2\to8}$$

o) Number of groups from which neutrons elastically scatter
p) Number of groups to which neutrons elastically scatter
q) Microscopic, P_o and P_1, elastic scatter, cross sections in the following order:

$$\sigma^{P_o}_{1\to1} \; ; \; \sigma^{P_1}_{1\to1} \; ; \; \sigma^{P_o}_{1\to2} \; ; \; \sigma^{P_1}_{1\to2}$$

$$\sigma^{P_o}_{2\to2} \; ; \; \sigma^{P_1}_{2\to2} \; ; \; \sigma^{P_o}_{2\to3} \; ; \; \sigma^{P_1}_{2\to4}$$

$$\vdots$$

$$\sigma^{P_o}_{11\to11} \; ; \; \sigma^{P_1}_{11\to11} \; ; \; \sigma^{P_o}_{11\to th} \; ; \; \sigma^{P_1}_{11\to th}$$

Figure 2: Input Library, Carbon Cross Sections

```
 URANIUM235 116 1 1 .0085 704. 689. 21582. 15. .9972 1442.5
[.1591E+01 .1204E+01 .1303E+01 .1305E+01 .1255E+01 .1213E+01
 .1598E+01 .4071E+01 .2310E+02 .7208E+02 .8172E+02][7   E+01]
[.8   E+01][.6272E-03 .2387E-01 .1738      .4312      .5266
 .4077      .4232      .1293E-01 .3459E-04 .5427E-02 .8050E-01
 .3119      .5108      .4773      .5816      .2013E-01 .5497E-04
 .0         .2376E-01 .1745      .4161      .4959      .7476
 .3038E-01 .8512E-04 .0         .0         .6236E-01 .2587
 .4155      .8215      .4055E-01 .1175E-03 .6023E-08 .0
 .0         .1156      .3104      .8789      .5527E-01 .1671E-03
 .1955E-07 .0         .0         .1965      .1032E+01 .8628E-01
 .2752E-03 .1633E-07 .0         .0         .6694      .1779
 .7098E-03 .2034E-06 .0         .0][.11  E+02][.1   E+01]
[.3130E+01 .1062E+02 .9810E-02 -.1247E+01 .5203E+01 .1733E+02
 .2267E-01 -.4294E+01 .4619E+01 .1191E+02 .3605E-01 -.2924E+01
 .3860E+01 .6376E+01 .3752E-01 -.9214      .3881E+01  .5201E+01
 .4281E-01 -.1068E+01 .4374E+01 .4665E+01 .5143E-01 -.5365
 .7348E+01 .3865E+01 .2801E-01 -.8345E-01 .1043E+02 .1182E+01
 .2631E-01 -.2356E-01 .1048E+02 .1077      .2195E-01 -.1834E-01
 .1241E+02 .1308      .2710E-01 -.2493E-01 .1385E+02 .2227
 .1167    -.1038]

[.5753E+01 .3828E+01 .3786E+01 .3528E+01 .3247E+01 .3028E+01
 .3728E+01 .7827E+01 .3768E+02 .1012E+03 .1701E+03]
```

Code:

a) → q) Same as in Figure 2

r) Microscopic, fast, absorption cross sections, σ_a^1, σ_a^2, . . . σ_a

s) Microscopic, fast $\nu\sigma_f^1$ in the order $\nu\sigma_f^1$, $\nu\sigma_f^2$

Note: These cross sections must not be included for chain one but must be included for chain two.

t) (Not shown) Number of groups in which (n,2n) occurs

u) (Not shown) Number of groups to which (n,2n) scatter occurs.

v) (n,2n) microscopic cross sections in same order as inelastic

Note: t), u), v) would appear in that order just before entry q).

Figure 3: Input Library, U235 Cross Sections

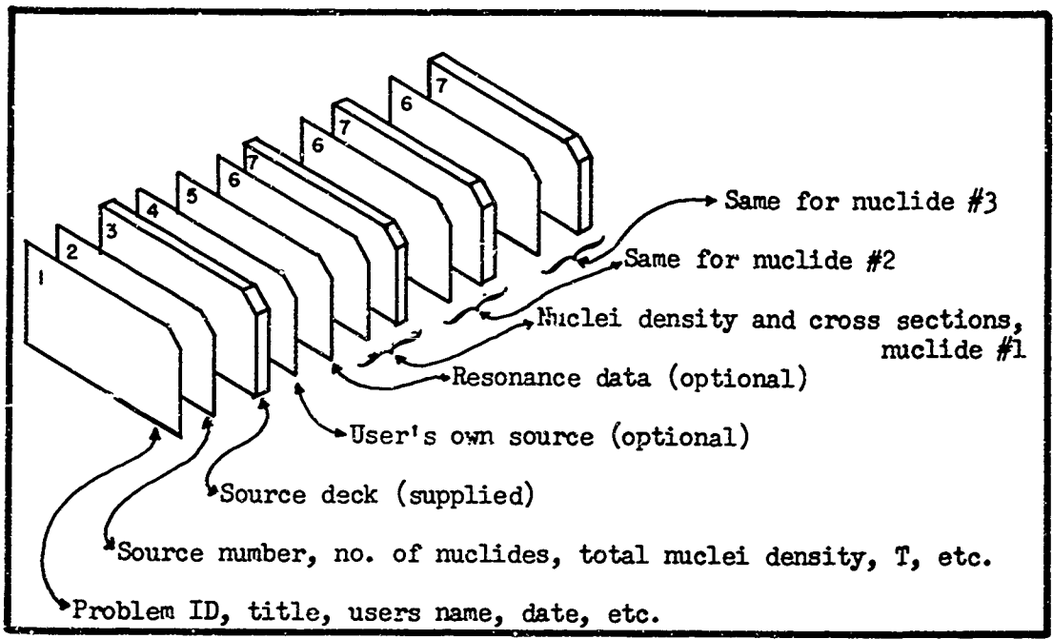

FIGURE 4: CHAIN ONE INPUT.

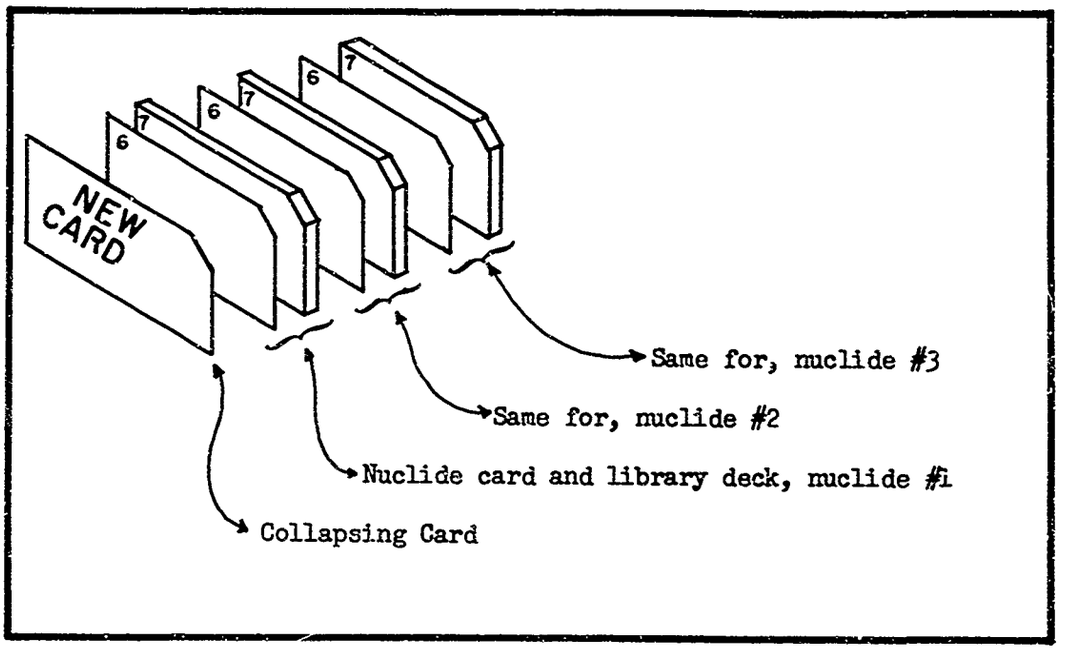

FIGURE 5: CHAIN TWO INPUT.

or with the sum of those on several card no. 6's if there is more than one absorbing nuclide.

Preparation of Input Data Chain, Two

Chain Two of Old Barnyard requires only one additional data card. This card bears up to twelve numbers. The first is the number of broad groups for which cross sections are to be calculated. The remaining numbers are the numbers of the lowest fine groups in the selected broad groups. The last of these should normally be eleven indicating that the last broad group extends down to .414 ev. Thus if the first number is five there will be five more numbers on the card. If the first number is 11, there will be 11 more numbers on the card.

Examples:

5 2 4 6 8 11

11 1 2 3 4 5 6 7 8 9 10 11

This additional card is the first input card for chain two. It is followed by the collection of cards 6 and 7 from chain one as shown in Figure 5.

Output

A sample output of the code is shown in Figure 6. This output is for a homogenized mixture representing an MTR type core. Figure 6a shows the output of chain one. Figure 6b shows the flux plot. Figures 6c, d and e are the chain two output. In this example, the user has asked for three group macroscopic cross sections which have boundries at fine groups 5 and 17.

Experience

The code in the form presented in Apendix A has been used routinely by several first year graduate students during the past academic term. These students had access to a set of instructions essentially equivalent to those contained here, and have used the code successfully without any other tutoring.

The code executes in about 15 minutes on the authors' IBM 1620. At least five minutes of this time is involved in computing the flux plot. However, this plot feature is believed to be very worthwhile from an educational point of view. Actually seeing the energy dependent flux for a particular problem affords the student considerable insight into the physics of that problem.

Status

There are several minor bugs in the program as presented in Appendix A. The most annoying of these is failure of the plot routine for a fast mixture (say pure uranium). Another disconcerting feature is that

$$\sigma_{tr} \neq \sigma_a + \sigma_{in} + \sigma P_o - \sigma P_1$$

This is probably due to the reciprocal formulation of σ_{tr}.

These problems are being corrected now and in addition better temperature and most probable thermal velocity calculations are being incorporated. Also, it is planned to incorporate simple fast fission factor and hetrogeneous cell calculations.

The final code along with a comprehensive theory and operating instructions will be published as the next and final report.

NETF CORE COMPOSITION FOR DB

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

URANIUM235	.00137621
URANIUM 38	.00015252
HYDROGEN	.48262000
OXYGEN	.24131000
ALUMINUM	.27583353

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	5.4905E-01
2	1.00	3.6788E+06	1.0824E-01	2.2786E+00
3	1.50	2.2313E+06	2.1044E-01	4.9358E+00
4	2.00	1.3534E+06	2.3139E-01	5.5195E+00
5	2.50	8.2085E+05	1.8048E-01	4.8660E+00
6	3.00	4.9787E+05	1.1483E-01	4.2230E+00
7	5.00	6.7379E+04	1.2439E-01	2.2664E+00
8	8.00	3.3546E+03	7.2094E-03	1.2042E+00
9	12.00	6.1442E+01	.0000E+00	1.0533E+00
10	16.00	1.1254E+00	.0000E+00	1.0000E+00
11	17.00	4.1399E-01	.0000E+00	1.0266E+00
12	17.50	2.5110E-01	.0000E+00	1.1047E+00
13	18.00	1.5230E-01	.0000E+00	1.4014E+00
14	18.50	9.2374E-02	.0000E+00	1.8501E+00
15	19.00	5.6028E-02	.0000E+00	1.8196E+00
16	19.50	3.3983E-02	.0000E+00	1.2708E+00
17	20.00	2.0612E-02	.0000E+00	6.8993E-01
18	20.50	1.2502E-02	.0000E+00	3.1839E-01
19	21.00	7.5826E-03	.0000E+00	1.3300E-01
20	21.50	4.5991E-03	.0000E+00	5.2364E-02

AGE TO INDIUM RESONANCE (1.46EV) IS	4.3711E+01	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	4.3970E+01	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	3.2857E+00	CM2
TOTAL MIGRATION AREA IS	4.7256E+01	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	2.8887E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	3.8368E+03	M/SEC
ABSORPTION PARAMETER IS	1.1892E+00	
SCATTERING PER RESONANCE ATOM IS	6.f078E+04	BARNS
EFFECTIVE RESONANCE INTEGRAL IS	2.5021E+02	BARNS

Figure 6a: Output, Chain One

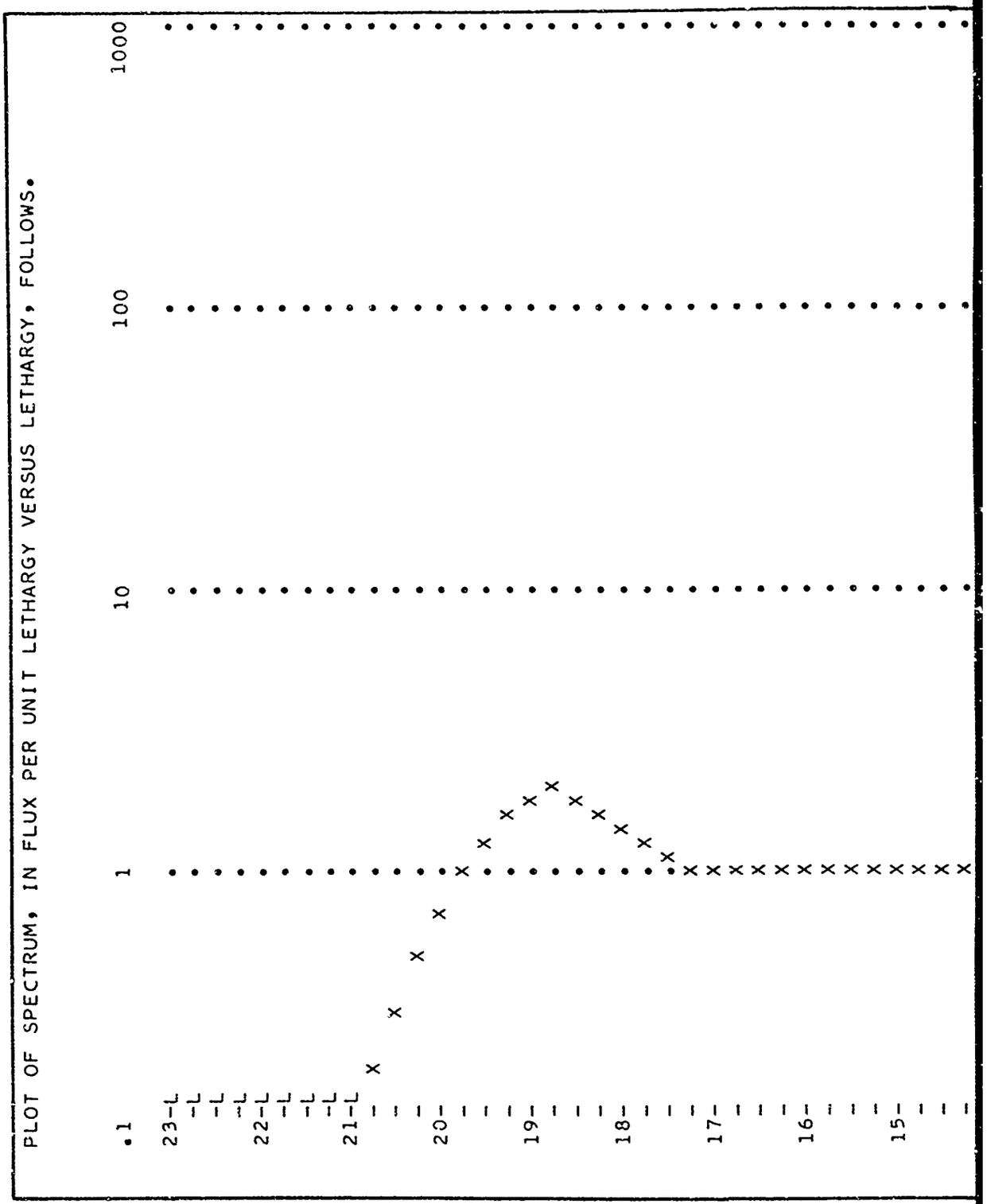

Figure 6b: Output, Flux Plot

A.

B.

ج.

NUCLIDE IS URANIUM235

ITS NUMBER DENSITY IS 1.1640E-04 PER BARN-CM

MACROSCOPIC CROSS SECTIONS, IN PER CM, FOLLOW.

GROUP	NUSIGF	SIGA	SIGR	SIGTR	SOURCE	LETHARGY
1	.000416	.000159	.000159	.000157	2.730272	5.000000
2	.010673	.005873	.005873	.001753	.007209	17.000000
3	.096155	.045928	.045928	.001746	.000000	99.999900

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.0000	.0000	.0000	.0000	.0000
1	2	.0000	.0000	.0000	.0000	.0000
1	3	.0000	.0000	.0000	.0000	.0000
2	2	.0000	.0000	.0000	.0000	.0000
2	3	.0000	.0000	.0000	.0000	.0000
3	3	.0000	.0000	15.0000	.0420	15.0000

NUCLIDE IS URANIUM 38

ITS NUMBER DENSITY IS 1.2900E-05 PER BARN-CM

MACROSCOPIC CROSS SECTIONS, IN PER CM, FOLLOW.

GROUP	NUSIGF	SIGA	SIGR	SIGTR	SOURCE	LETHARGY
1	.000008	.000004	.000005	.000073	2.730272	5.000000
2	.000000	.000002	.000002	.000149	.007209	17.000000
3	.000000	.000057	.000057	.000107	.000000	99.999900

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.0000	.0000	.0001	.0000	.0001
1	2	.0000	.0000	.0000	.0000	.0000
1	3	.0000	.0000	.0000	.0000	.0000
2	2	.0000	.0000	.0002	.0000	.0002
2	3	.0000	.0000	.0000	.0000	.0000
3	3	.0000	.0000	8.3000	.0232	8.3000

Figure 6c: Output, Chain Two

NUCLIDE IS HYDROGEN

ITS NUMBER DENSITY IS 4.082CE-02 PER BARN-CM

MACROSCOPIC CROSS SECTIONS, IN PER CM, FOLLOW.

GROUP	NUSIGF	SIGA	SIGR	SIGTR	SOURCE	LETHARGY
1	.000000	.000000	.054385	.050234	2.730272	5.000000
2	.000000	.001101	.208083	.273940	.007209	17.000000
3	.000000	.007714	.007714	.525538	.000000	99.999900

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.0000	.0000	.1552	.1152	.1552
1	2	.0000	.0000	.0544	.0231	.0544
1	3	.0000	.0000	.0000	.0000	.0000
2	2	.0000	.0000	.6116	.4365	.6116
2	3	.0000	.0000	.2070	.1081	.2070
3	3	.0000	.0000	38.0000	25.1332	38.0000

NUCLIDE IS OXYGEN

ITS NUMBER DENSITY IS 2.4100E-02 PER BARN-CM

MACROSCOPIC CROSS SECTIONS, IN PER CM, FOLLOW.

GROUP	NUSIGF	SIGA	SIGR	SIGTR	SOURCE	LETHARGY
1	.000000	.000206	.001271	.054872	2.730272	5.000000
2	.000000	.000000	.003552	.084439	.007209	17.000000
3	.000000	.000003	.000003	.096959	.000000	99.999900

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.0000	.0000	.0766	.0080	.0766
1	2	.0000	.0000	.0011	-.0003	.0011
1	3	.0000	.0000	.0000	.0000	.0000
2	2	.0000	.0000	.0846	.0047	.0846
2	3	.0000	.0000	.0036	-.0011	.0036
3	3	.0000	.0000	4.2000	.1751	4.2000

Figure 6d: Output, Chain Two, Continued

NUCLIDE IS ALUMINUM

ITS NUMBER DENSITY IS 2.3330E-02 PER BARN-CM

MACROSCOPIC CROSS SECTIONS, IN PER CM, FOLLOW.

GROUP	NUSIGF	SIGA	SIGR	SIGTR	SOURCE	LETHARGY
1	.000000	.000073	.001230	.057909	2.730272	5.000000
2	.000000	.000527	.001230	.032878	.007209	17.000000
3	.000000	.003073	.003073	.031525	.000000	99.999900

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.0044	.0000	.0768	.0184	.0812
1	2	.0000	.0000	.0012	-.0004	.0012
1	3	.0000	.0000	.0000	.0000	.0000
2	2	.0000	.0000	.0325	.0010	.0325
2	3	.0000	.0000	.0007	-.0002	.0007
3	3	.0000	.0000	1.4000	.0518	1.4000

MAXWELL-BOLTZMAN FACTOR = 1.123, [illegible]-WIGNER = 1.7462097E+00

Figure 6e: Output, Chain Two, Continued

BIBLIOGRAPHY

1. Argonne National Laboratory. Reactor Physics Constants. USAEC Document ANL-5800, Lemont, Illinois, July 1963.

2. Joanou, G. D., and J. S. Dudek. GAM-I Consistent P_1 Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants. USAEC Document GA-1850. John Jay Hopkins Laboratory for Pure and Applied Science, San Diego, California; 1961.

3. Murray, R. L., Nuclear Reactor Physics, page 63, Prentice-Hall Inc., Englewood Cliffs, N. J., 1957.

4. Hansen, Robert H., Theoretical Calculation of Neutron Age Defense Documentation Center document AD 603603, Thesis, Air Force Institute of Technology, Dayton, Ohio, 1964.

5. Murray, R. L., ibid, page 61.

6. Isbin, H.S., Introductory Reactor Theory. page 459, Reinhold Publishing Corp., New York, N.Y. 1963.

7. Weinberg, A.M., and E. D. Wigner, The Physical Theory of Neutron Chain Reactors, page 337. The University of Chicago Press, Chicago, Illinois, 1958.

8. Cohen, E.R. "The Neutron Velocity Spectrum in a Heavy Moderator," Nuclear Science and Engineering 2:227-245, May 1957.

9. Sims, E. P. The Thermal Neutron Spectrum in a Finite-Mass Medium. Defense Documentation Center document AD621119, Thesis, Air Force Institute of Technology, Dayton, Ohio, 1965.

10. Field, J. A. A.[1], D. A. Jardine[2], E. S. Lee[1], J. A. N. Lee[3], and D. G. Robinson[2]. Kingston Fortran II Language Specifications. Document distributed by the authors, July 15, 1964.

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APPENDIX A

```
$       JOB  OLD BARNYARD  CHAIN 1, MAY 1966
C       THIS CARD ORIGINS THE PROGRAM AT ADDRESS 10970 IN MEMORY.  IN ORDER
C       TO DO THIS YOU MUST USE A SPECIAL SET OF SHORT SUBROUTINES.
$       ORIGIN 10970
C       STORING IN COMMON THOSE DATA USED IN CHAIN 2 OR IN SUBPROGRAMS
        COMMON LGP(13),KE,FLX(11),U(11),SOURCE(11),DENBR,NRNUC,TEMPTR,XBAR
        DIMENSION SA(11),STR(11),SR(11),XSAM(168),SSN(12,13),S2N(12,13),X(
       111),T(12)

C       READING IN THE FIRST TWO DATA CARDS
 1001   READ 1060,LGP
 1060   FORMAT(13A5)
        READ,NEW,NRNUC,DENBR,TEMPTR,RES
        PUNCH1060,LGP
        PUNCH1065
 1065   FORMAT(//)
        TYPE1060,LGP

C       READING IN THE DECK OF SOURCES
        READ 1070,U
        K=0
        DO 1046 J=1,10
        IF(K)1045,1043,1045
 1043   READ1060,LGP
        READ1070,INDEX,SOURCE
 1070   FORMAT(12N)
        IF(INDEX-NEW)1046,1044,1046
 1044   K=1
        GO TO 1046
 1045   READ 1060,PU
        READ,PU
 1046   CONTINUE
        IF(NEW-10)1050,1049,1047

C       SETTING THE UNIT SOURCE IN ITS GROUP (IF APPLICABLE)
 1047   SOURCE(NEW-10)=1.
        LGP(10)=NEW-10
        PUNCH 1071,LGP
 1071   FORMAT(9A5,I3,3A5)
        GO TO 1048

C       READING IN THE USERS OWN SOURCE (IF APPLICABLE)
 1049   READ1070,SOURCE
 1050   PUNCH 1060,LGP
 1048   TYPE 600
 600    FORMAT(/16HBEGIN EPITHERMAL)
        PUNCH 1058,TEMPTR
 1058   FORMAT(/22HSYSTEM TEMPERATURE IS ,F6.1,7H KELVIN)
        PUNCH 1059
 1059   FORMAT(/18HATOM FRACTIONS ARE/)

C       CLEARING THE MATRICES OF ANY PREVIOUS PROBLEM
        DO 1000 J=1,11
        STR(J)=0.
        SR(J)=0.
```

```
 1000 SR(J)=.0
      DO 1002 J=1,12
      DO 1002 K=1,13
      SSN(J,K)=0.
 1002 S2N(J,K)=0.

C     READING IN THE RESONANCE DATA CARD (IF APPLICABLE)
      IF(RES)995,994,995
  995 READ,JJJ,PU,RIEFF
      PU=PU/DENBR

C     SETTING UP THE DO LOOP FOR EACH OF THE NUCLIDES
  994 PROB=1.
      KKK=0
      PSIGF=0.
      PSIGM=0.
      PSIGO=0.
      SIGTT=.0
      SIGDEL=.0
      SIGSTH=0.
      DO 1020 I=1,NRNUC

C     READING IN THE NUCLIDE DATA CARDS
      READ 1061, SAM, SAMM, ATOMS, P, MOD, DENS
 1061 FORMAT(3A5,9N)
      P=P/DENBR
      READ 1061,SAM,SAMM,ATOMS,LS,NA,LQ,PSI,SIGT,SIGA,DELTA,SIGS
      PUNCH1066,SAM,SAMM,ATOMS,P
 1066 FORMAT(3A5,F15.8)

C     CALCULATING TOTAL THERMAL SCATTERING SIGMA
      SIGSTH=SIGSTH+P*SIGS

C     CALCULATING THERMAL UTILIZATION AND ABSORPTION PARAMETER
      GO TO (993,992,996),MOD
  992 PSIGF=PSIGF+P*SIGA
      GO TO 991
  996 PSIGO=PSIGO+P*SIGA
      GO TO 991
  993 PSIGM=PSIGM+P*SIGA
      SIGDEL=SIGDEL+SIGT*P*DELTA
      SIGTT=SIGTT+P*SIGT

C     READING IN THE WHOLE CROSS SECTION LIBRARY FOR THE NUCLIDE
  991 READ 1067,(XSAM(J),J=1,LS)
 1067 FORMAT(6N)
      KZ=0
      KKZ=0

C     CHECKING FOR ABSORPTION
      IF(NA)1006,1006,1004
 1004 DO1005 J=1,11
      KZ=KZ+1
      STR(J)=STR(J)+XSAM(KZ)*P
 1005 SR(J)=SR(J)+XSAM(KZ)*P
```

```
 1006 KZ=KZ+1
      IN=XSAM(KZ)
      KZ=KZ+1
      IIN=XSAM(KZ)
      DO 1017 J=1,IN
      KK=J+IIN
      IF(KK-12)1008,1008,1007
 1007 KK=12
 1008 DO 1017 K=J,KK
      GO TO (1009,1010,1009,1011),LQ
 1009 KZ=KZ+1

C     INELASTIC CROSS SECTION, GROUP TO GROUP
      SSN(J,K+1)=SSN(J,K+1)+XSAM(KZ)*P
      GO TO 1012
 1010 KZ=KZ+1

C     N-2N CROSS SECTIONS, GROUP TO GROUP
      S2N(J,K+1)=S2N(J,K+1)+XSAM(KZ)*P
      GO TO 1012
 1011 IF(KKZ)2001,2000,2001
 2000 KZ=KZ+1
      KKZ=1
      GOTO2002
 2001 KZ=KZ+2

C     PUTTING ELASTIC (P0) AND ELASTIC (P1) IN THE BOTTOM OF SSN AND S2N
 2002 LS=13-J
      LE=13-K
      SSN(LS,LE)=SSN(LS,LE)+XSAM(KZ)*P
      S2N(LS,LE)=S2N(LS,LE)+XSAM(KZ+1)*P/3.
 1012 STR(J)=STR(J)+XSAM(KZ)*P
      IF(K-J)1016,1013,1016
 1013 GO TO (1017,1014,1017,1015),LQ
 1014 SR(J)=SR(J)-XSAM(KZ)*P
      GO TO 1017
 1015 STR(J)=STR(J)-XSAM(KZ+1)*P/3.
      GO TO 1017
 1016 SR(J)=SR(J)+XSAM(KZ)*P
 1017 CONTINUE
      GO TO(1018,1018,1019,700),LQ
 1018 LQ=4
      GO TO 1006
 1019 LQ=2
      GO TO 1006

C     READING IN THE GRID BACKGROUND FOR THE SPECTRUM PRINTOUT
  700 READ 2222,(XSAM(J),J=41,120)
 2222 FORMAT(80A1)

C     CHECKING FOR RESONANCE CALCULATION
      IF(KKK)705,701,705
  701 IF(RES)704,1020,704
  704 KKK=1
```

```
C       CALCULATING PSI*SIGMA-S FOR THE MIXTURE
        TSIG8=0.
        TSIG9=0.
        PSIG8=0.
 705    SIG8=SSN(5,5)+SSN(5,4)+SSN(5,3)+SSN(5,2)+SSN(5,1)-TSIG8
        TSIG8=SSN(5,5)+SSN(5,4)+SSN(5,3)+SSN(5,2)+SSN(5,1)
        SIG9 =SSN(4,4)+SSN(4,3)+SSN(4,2)+SSN(4,1)-TSIG9
        TSIG9=SSN(4,4)+SSN(4,3)+SSN(4,2)+SSN(4,1)
        PSIG8=PSIG8+PSI*P*(SIG8+SIG9)*.5
 1020 CONTINUE

C       ALL CROSS SECTIONS HAVE BEEN CALCULATED AT THIS POINT

        IF(KKK)720,730,720
 720    TYPE 601
 601    FORMAT(/15HBEGIN RESONANCE)

C       CALCULATING SCATTERING PER RESONANCE ATOM (IF APPLICABLE)
        SCAT=.5*(TSIG8+TSIG9)/PU

C       SELECTING THE EMPIRICAL EQUATION FOR RESONANCE INTEGRAL
        IF (RIEFF)728,721,728
 721    GO TO(722,725),JJJ
 722    IF(SCAT-4000.)723,723,724
 723    RIEFF=2.69*SCAT**.471
        GO TO 728
 724    RIEFF=EXPF(5.64-163./(SCAT**.65),
        GO TO 728
 725    IF(SCAT-4500.)726,726,727
 726    RIEFF=8.33*SCAT**.253
        GO TO 728
 727    RIEFF=70.

C       CALCULATING RESONANCE ESCAPE PROBABILITY
 728    PROB=EXPF(PU*RIEFF/PSIG8)

C       ADJUSTING GROUP 8 REMOVAL CROSS SECTION FOR RESONANCE ESCAPE
        SR(8)=SR(8)*PROB
        STR(8)=SR(8)+SSN(5,5)-S2N(5,5)

C       HANSENS FINITE DIFFERENCE EQUATIONS FOR EPITHERMAL FLUX AND AGE
 730    SAMM=1./(3.*DENBR*DENBR)
        PUNCH 1051
 1051 FORMAT(//57X,14HRELATIVE GROUP/5HGROUP,7X,8HLETHARGY,7X,10HENERGY,
     1 EV,8X,6HSOURCE,6X,14HFLUX,  /UNIT U/)
        RHO=.0
        DO 1029 K=1,11
        LE=13-K
        DENS=SAMM/(STR(K)*SR(K))
        ATOMS=0.
        TOTWT=0.
        P=0.
        T(K)=0.
        FLX(K)=0.
        SA(K)=0.
```

```
      IF(K-1)1027,1027,1025
 1025 DO 1026 J=1,K-1
      LS=13-J
      SAM=FLX(J)*(SSN(LS,LE)+SSN(J,K+1)+2.*S2N(J,K+1))/SR(J)
      ATOMS=ATOMS+SAM
      TOTWT=TOTWT+SAM*T(J)
 1026 P=P+S2N(LS,LE)*SA(J)
 1027 FLX(K)=ATOMS+SOURCE(K)
      SA(K)=FLX(K)*DENS+P/STR(K)
      T(K)=(SA(K)+TOTWT)/FLX(K)
      X(K)=U(K)-RHO
      RHO=U(K)
 1029 CONTINUE
      F=SR(10)*4./FLX(10)

C     CALCULATING TOTAL FLUX PER GROUP
      DO 1023 K=1,10
 1023 FLX(K)=FLX(K)/SR(K)

C     CALCULATING AND NORMALIZING FLUX PER UNIT LETHARGY
      DO 1022 K=1,10
      XSAM(40-K)=F*FLX(K)/X(K)
      KK=40-K
C     OUTPUTTING EPITHERMAL RESULTS
 1022 PUNCH 1052,K,U(K),1.0E+07*EXP(-U(K)),SOURCE(K),XSAM(KK)

C     NOTICE THAT TOTAL FLUX PER GROUP HAS BEEN LEFT IN STORAGE IN FLX(11)

C     CALCULATING AGE TO INDIUM RESONANCE
      SAMM=(1(10)-T(9))*0.935+T(9)

C     CHECKING FOR MODERATOR IN THE SYSTEM
      IF(PSIGM)1024,1021,1024
 1021 PUNCH 850,SAMM
      TYPE 1028
 1028 FORMAT(36HTHERE IS NO MODERATOR IN THE SYSTEM./36HNO THERMAL CALCU
     1LATION WILL BE DONE.)
      GO TO 860

C     STARTING THE THERMAL CALCULATION
 1024 TYPE 602
 602  FORMAT(/13HBEGIN THERMAL)

C     THE ABSORPTION PARAMETER
      DELTA=SIGDEL*(PSIGF+PSIGM+PSIGO)/(SIGTT*(PSIGM+PSIGO))
      AREAX=.0
      PREFX=.0
      PREVF=0.
      II=0
      AN2=1.
      P=0.

C     ARRANGING FOR INCREMENTS OF 0.25 LETHARGY UNITS
      ETODU=EXP(.125)-1.
```

```
C       Y HERE EQUALS X.  SINCE X WAS SUBSCRIPTED ABOVE
        Y=EXP(-11.5)*SQRT(1.0E+12/(8.616*TEMPTR))
        I=0

C       DOING THE SERIES SOLUTION TO WILKINS EQUATION
        AN1=AN2*DELTA*.333333
        DSUM=AN1
  802   SUM=AN2+AN1*Y
        XTON=Y
        AN3=0.
        N=4
  803   N=N+1
        XN=N-3
        IF (P)805,804,805

C       THE RECURSION RELATION FOR X=0
  804   AN=(2.*(XN-2.)*AN2+DELTA*AN1)/(XN*(XN+2.))
        GO TO 806

C       THE GENERAL RECURSION RELATION
  805   AN=(AN1*(XN-1.)*(XN+1.-2.*P*P)-AN2*(4.*P*(XN-2.)+DELTA)-AN3*2.*(X
       1N-3.))/(P*XN*(1.-XN))
        AN3=AN2
  806   DTERM=XN*AN*XTON
        XTON=XTON*Y
        TERM=AN*XTON
        SUM=SUM+TERM
        DSUM=DTERM+DSUM
        AN2=AN1
        AN1=AN

C       CHECKING FOR CONVERGENCE OF THE SERIES
        IF(ABS(TERM/SUM)-1.0E-06)807,807,803
  807   IF(ABS(DTERM/DSUM)-1.0E-06)808,808,803
  808   P=Y+P

C       FINDING N(X) FOR THE POINT X=P
        F=EXP(-P*P)*SUM*P**2

C       LOCATING THE PEAK OF THE N(X) CURVE
        IF(II)816,814,816
  814   IF(F-PREVF)815,815,816

C       CALCULATING THE MOST PROBABLE VELOCITY
  815   VPROB=SQRT(1.64775+04*TEMPTR)*P
        II=1
  816   PREVF=F

C       CALCULATING FLUX PER UNIT U WHICH IS X*X*N(X)
        F=F*P*P
        I=I+1

C       STORING THE VALUES OF THERMAL FLUX
        XSAM(I)=F
```

```
C       CALCULATING THE AREA UNDER THE XN(X) CURVE BY TRAPEZOIDAL RULE
        F=F/P
        AREAX=AREAX+(F+PREFX)*Y*.5
        PREFX=F
        IF(P-6.6)809,810,810
  809   AN2=SUM
        AN1=DSUM
        Y=P+P*ETODU
        Y=Y-P
        GO TO 802

C       CALCULATING THE AREA UNDER THE FLUX CURVE ANALYTICALLY
  810   C=2.*F*(1./(P*P)-1.)+P*P*EXP(-P*P)*DSUM
        AREA=(P*C+2.*(P*P-1.)*F/P)/DELTA

C       FINDING THE AVERAGE NORMALIZED VELOCITY
        XBAR=AREAX/AREA

C       NORMALIZING THE STORED THERMAL FLUXES
        F=1./XSAM(29)
        DO 812 J=1,29
  812   XSAM(J)=XSAM(J)*F
        P=16.5

C       OUTPUTTING THE THERMAL FLUXES
        DO 813 K=1,10
        P=P+.5
        J=27-K-K
  813   PUNCH 1052,K+10,P,1.0E+07*EXP(-P),0.,XSAM(J)
 1052   FORMAT(I3,F15.2,3X,1P3E16.4)

C       OUTPUTTING THE THERMAL CONSTANTS
        PUNCH 850,SAMM
        PUNCH 851,T(10)
  850   FORMAT(//5X,35HAGE TO INDIUM RESONANCE (1.46EV) IS,1PE20.4,4H CM2)
  851   FORMAT(/5X,36HAGE TO ARBITRARY THERMAL (1.12EV) IS, 1PE19.4,4H CM2
       1)

C       CALCULATING THE DIFFUSION LENGTH SQUARED AND MIGRATION AREA
        SIGATH=PSIGM+PSIGF+PSIGO
        ELSQRD=.33333*SIGSTH*XBAR/(SIGATH*(DENBR*(SIGSTH+SIGATH/XBAR))**2)
        PUNCH 852,ELSQRD
  852   FORMAT(/5X,35HTHERMAL DIFFUSION LENGTH SQUARED IS,1PE20.4,4H CM2)
        EMSQRD=T(10)+ELSQRD
        PUNCH 853,EMSQRD
  853   FORMAT(/5X,23HTOTAL MIGRATION AREA IS,1PE32.4,4H CM2)
        PUNCH 854,VPROB
  854   FORMAT(/5X,41HMOST PROBABLE THERMAL NEUTRON VELOCITY IS,1PE14.4,6H
       1 M/SEC)
        VBAR =SQRT(1.6477E+04*TEMPTR)*XBAR
        PUNCH 855,VBAR
  855   FORMAT(/5X,35HAVERAGE THERMAL NEUTRON VELOCITY IS,1PE20.4,6H M/SEC
       1)
        PUNCH 856,DELTA
  856   FORMAT(/5X,23HABSORPTION PARAMETER IS,1PE32.4)
```

```
C       OUTPUTTING THE RESULTS OF THE RESONANCE CALCULATION
        IF(PES)859,860,859
  859   PUNCH 857,SCAT
  857   FORMAT(/5X, 32HSCATTERING PER RESONANCE ATOM IS,1PE23.4,6H BARNS)
        PUNCH 858,RIEFF
  858   FORMAT(/5X,31HEFFECTIVE RESONANCE INTEGRAL IS,1PE24.4,6H BARNS)
  860   PUNCH 861
  861   FORMAT(78X,1H-)

C       WARNING THE USER TO SET SWITCH FOR SPECTRUM OUTPUT
        TYPE 862
  862   FORMAT(/52HPUT SWITCH 1 ON FOR SPECTRUM PRINTOUT.  PRESS START.)
        PAUSE
        IF(SENSE SWITCH 1)3000,2999

C       THE SPECTRUM PUNCHOUT SUBPROGRAM
 2999   TYPE 863
  863   FORMAT(/28HNO SPECTRUM WILL BE PUNCHED.)
        GO TO 1091
 3000   TYPE 603
  603   FORMAT(/14HBEGIN SPECTRUM)

C       STORING THE SYMBOLS FOR THE GRAPH
        X(1)=1HX
        X(2)=1HL
        X(3)=1HH
        PUNCH 300
  300   FORMAT(69HPLOT OF SPECTRUM, IN FLUX PER UNIT LETHARGY VERSUS LETHA
       1RGY, FOLLOWS.//)
        PUNCH 301

C       REDUCING FLUXES TO LOG SCALE 80 UNITS WIDE
        DO 3001 J=1,39
 3001   XSAM(J)=((LOG(XSAM(J))/2.30259)+1.)*20.+.5
        KK=9
        I=0

C       ONE PASS THROUGH THE LOOP FOR EACH LINE OF OUTPUT
        DO 3005 J=1,93
        IF(KK)3011,3014,3011

C       CHECKING FOR EPITHERMAL OR THERMAL FLUXES
 3011   IF(J-29)3009,3009,3012
 3012   PU=J
        P =23.2-PU/4.

C       ADJUSTING FOR DIFFERENT WIDTH GROUPS
        IF(P-U(KK))3013,3013,3014
 3013   KK=KK-1
 3014   K=XSAM(39-KK)
        GO TO 3008
 3009   K=XSAM(J)
 3008   I=I+1
```

```
C       NUMBERS FOR THE LETHARGY SCALE
        LE=23-J/4

C       CHECKING FOR VALUES TOO LARGE FOR SCALE
        IF(K-79)3015,3016,3006
 3006   N=3
        K=79
        GO TO 3004

C       CHECKING FOR VALUES TOO SMALL  FOR SCALE
 3015   IF(K-3)3019,3017,3016
 3017   GO TO (3018,3016,3016,3016,3018),I
 3018   N=2
        K=4
        GO TO 3004
 3019   IF(K-2)3018,3017,3017
 3016   N=1
 3004   KZ=K+39
        KKZ=K+41
        GO TO(3002,3003,3003,3003,3002),I

C       OUTPUTTING THE SPECTRUM GRAPH
 3002   PUNCH 302,LE,(XSAM(LS),LS=43,KZ),X(N),(XSAM(LS),LS=KKZ,120)
 302    FORMAT(I2,78A1)
        I=1
        GO TO 3005
 3003   PUNCH 2222,(XSAM(LS),LS=41,KZ),X(N),(XSAM(LS),LS=KKZ,120)
 3005   CONTINUE
        PUNCH 301
 301    FORMAT(/2H.1,17X,1H1,18X,2H10,18X,3H100,15X,4H1000/)
 1091   TYPE 1092
 1092   FORMAT(/14HEND OF CHAIN 1//39HTO COLLAPSE CROSS SECTIONS LOAD CHAI
       1N 2/)

C       THIS PUNCH MAKES THE 407 LISTER ADVANCE A SHEET OF PAPER
        PUNCH 861

C       THIS PUNCH AVOIDS RUNNING OUT THE LAST PUNCHED OUTPUT CARD
        PUNCH 1065
        STOP
        END
$       EOJ
```

```
C       THIS IS PART II OF OLD BARNYARD.  CROSS SECTION COLLAPSING.


$       JOB  OLD BARNYARD CHAIN 2
$       ORIGIN 10970

C       NOTICE COMMON STATEMENT IS SAME AS IN CHAIN 1
        COMMOM LGP(13),KE,FLX(11),U(11),SOURCE(11),DENBR,NRNUC,TEMPTR,XBAR
        DIMENSION SA(11),STR(11),SR(11),XSAM(168),SSN(12,13),S2N(12,13),
       1SN(12,13)
        TYPE 604
 604    FORMAT(/16HBEGIN COLLAPSING)

C       READING IN NUMBER OF BROAD GROUPS AND THEIR BOUNDARY FINE GROUPS
        READ 1070,NBG,(LGP(J),J=2,NBG+1)
 1070   FORMAT(12N)
        MXL=NBG+1
        LGP(1)=0
        LGP(NBG+2)=12

C       BETTER LOOK NOW AT FUNCTION SUBROUTINE KS AT END OF PROGRAM

C       NORMALIZING THE FINE GROUP FLUXES IN EACH BROAD GROUP
C       AND ADDING UP THE SOURCE IN EACH BROAD GROUP
        DO 2 J=1,NBG
        SA(J)=0.
        DO  2 K=KS(J),KE
 2      SA(J)=SA(J)+FLX(K)
        DO 4 J=1,NBG
        SAM=.0
        DO  3 K=KS(J),KE
        SAM=SAM+SOURCE(K)
 3      FLX(K)=FLX(K)/SA(J)
 4      SOURCE(J)=SAM

C       ONE TRIP THORUGH THE DO LOOP FOR EACH NUCLIDE
        DO 55 N=1,NRNUC

C       CLEARING THE MATRICES OF ANY PREVIOUS PROBLEM
        DO 555 J=1,11
        SA(J)=0.
        SR(J)=0.
 555    STR(J)=0.
        DO 6 J=1,12
        DO 6 K=1,13
        SSN(J,K)=0.
 6      S2N(J,K)=0.

C       READING IN THE NUCLIDE DATA CARDS
        READ 1061,SAM,SAMM,ATOMS,P,MOD,DENS
 1061   FORMAT(3A5,9N)
        P=P/DENBR
        READ 1061,SAM,SAMM,ATOMS,LS,NA,LG,PSI,SIGT,SIGA,DELTA,SIGS,TERMU
        PUNCH 200,SAMM,ATOMS,P*DENBR
 200    FORMAT(//11HNUCLIDE IS ,2A5,///25H     ITS NUMBER DENSITY IS,1PE11.
```

```
     1H,12H PER BARN-CM//)

C      CHECKING FOR MICRO OR MACROSCOPIC CROSS SECTIONS
       IF(DENS-1.)61,62,61
 61    DENS=P*DENBR
       PUNCH 202
 202   FORMAT(46HMACROSCOPIC CROSS SECTIONS, IN PER CM, FOLLOW.)
       GO TO 49
 62    PUNCH 201
 201   FORMAT(45HMICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.)
 49    PUNCH 220
 220   FORMAT(//79HGROUP      ABSORPTION     REMOVAL     TRANSPORT    DIFF CO
     1EF     SOURCE      LETHARGY/)

C      READING IN THE WHOLE CROSS SECTION LIBRARY FOR THE NUCLIDE
       READ 1067,(XSAM(J),J=1,IS)
 1067  FORMAT(5N)
       KZ=0

C      CHECKING FOR ABSORPTION
       IF(NA)10,10,7
 7     DO 9 J=1,NBG
       DO  8 K=KS(J),KE
       KZ=KZ+1
       SA(J)=SA(J)+XSAM(KZ)*FLX(K)*DENS
 8     STR(K)=XSAM(KZ)
 9     SR(J)=SA(J)
 10    DO 12 J=1,12
       DO 12 K=1,13
 12    SN(J,K)=0.
       KZ=KZ+1
       IN=XSAM(KZ)
       KZ=KZ+1
       IIN=XSAM(KZ)
       DO 15 J=1,IN
       KK=J+IIN
       IF(KK-12)14,14,13
 13    KK=12
 14    DO 15 K=J,KK
       GO TO (141,141,141,371),LQ

C      USING SN AS A TEMPORARY STORAGE MATRIX FOR INELASTIC OR N-2N
 141   KZ=KZ+1
       STR(J)=STR(J)+XSAM(KZ)
       SN(J,K+1)=XSAM(KZ)*FLX(J)*DENS
       GO TO 15

C      USING SN AS A TEMPORARY STORAGE MATRIX FOR ELASTIC (P0) AND (P1)
 371   NA=13-J
       KE=13-K
       KZ=KZ+1
       XSAM(KZ+1)=XSAM(KZ+1)*.33333
       STR(J)=STR(J)+XSAM(KZ)-XSAM(KZ+1)
       SN(J,K+1)=XSAM(KZ)*FLX(J)*DENS
       SN(NA,KE)=XSAM(KZ+1)*FLX(J)*DENS
```

```
      KZ=KZ+1
 15   CONTINUE
      DO 19 J=1,NBG
      DO 19 K=J,NBG+1
      LS=LGP(K)+1
      LE=LGP(K+1)
      DO 19 I=KS(I),KE
      IF(LS-I)16,17,17
 16   LS=I
 17   DO 19 L=LS,LE
      GO TO (11,22,11,34),LQ

C     INELASTIC CROSS SECTION, GROUP TO GROUP
 11   ISN(J,K+1)=SSN(J,K+1)+SN(I,L+1)
      GO TO 99

C     ELASTIC CROSS SECTION, GROUP TO GROUP (P0) AND (P1)
 34   NA=13-J
      KK=13-K
      IN=13-I
      IIN=13-L
      SSN(NA,KK)=SSN(NA,KK)+SN(I,L+1)
      S2N(NA,KK)=S2N(NA,KK)+SN(IN,IIN)
 99   IF(J-K)18,19,19

C     N-2N CROSS SECTION, GROUP TO GROUP
 22   S2N(J,K+1)=S2N(J,K+1)+SN(I,L+1)
      IF(J-K)18,30,18
 30   SR(J)=SR(J)-SN(I,L+1)
      GO TO 19
 18   SR(J)=SR(J)+SN(I,L+1)
 19   CONTINUE
      GO TO (91,91,92,421),LQ
 91   LQ=4
      GO TO 10
 92   LQ=2
      GO TO  10

C     CALCULATING BROAD GROUP SIGMA TRANSPORT
 421  DO 46 J=1,NBG
      SAM=0.
      DO 45 K=KS(J),KE

C     NOTICE RECIPROCAL RELATIONSHIP
 45   SAM=SAM+FLX(K)/STR(K)
 46   STR(J)=DENS/SAM

C     OUTPUTTING SINGLE GROUP VALUES
      DO 50 J=1,NBG
      K=J+1
 50   PUNCH 102,J,SA(J),SR(J),STR(J),.33333/STR(J),SOURCE(J),U(LGP(K))

C     CALCULATING AND OUTPUTTING THERMAL GROUP VALUES
      SIGA=SIGA/XBAR
      SIGTR=SIGS*TERMU+SIGA
```

```
      PUNCH 102,MXL,SIGA,SIGA,SIGTR,.33333/SIGTR,0.,99.9999
  102 FORMAT(I3,3X,6F12.4)

C     READING AND IGNORING THE GRID BACKGROUND CARD IN EACH LIBRARY DECK
      READ 1061,SAM
      PUNCH 205
  205 FORMAT(//7HSCATTER,37X,7HELASTIC,7X,7HELASTIC,8X,5HTOTAL/8HFROM  T
     1O,7X,9HINELASTIC,8X,4HN-2N,10X,4H(P0),10X,4H(P1),7X,8HTRANSFER/)

C     OUTPUTTING GROUP TO GROUP VALUES
      DO 54 J=1,NBG
      DO 54 K=J,MXL
      LS=13-J
      LE=13-K
      SAM       =SSN(J,K+1)+2.*S2N(J,K+1)+SSN(LS,LE)
   54 PUNCH103,J,K,SSN(J,K+1),S2N(J,K+1),SSN(LS,LE),S2N(LS,LE),SAM

C     OUTPUTTING THERMAL GROUP TO GROUP VALUES
   55 PUNCH 103,MXL,MXL,0.,0.,SIGS,(1.-TERMU)*SIGS,SIGS
  103 FORMAT(I3,I5,5F14.4)
 1093 TYPE 1063
 1063 FORMAT(/15HEND OF PROGRAM./)
      PUNCH 1065
 1065 FORMAT(//1X)
      STOP
      END

C     A FUNCTION SUBROUTINE TO SAVE A LOT OF WRITING ON DO LOOPS
      FUNCTION KS(J)
      COMMOM LGP(13),KE
      KS=LGP(J)+1
      KE=LGP(J+1)
      RETURN
      END
$     EOJ
```